ACCESSION #: 9310210020
LICENSEE EVENT REPORT (LER)

FACILITY NAME: LaSalle County Station Unit 1 PAGE: 1 OF 17

DOCKET NUMBER: 05000373

TITLE: Unit 1 Scram and Loss of Off-Site Power Due to Bus Duct
Water Intrusion
EVENT DATE: 09/14/93 LER #: 93-015-00 REPORT DATE: 10/12/93

OTHER FACILITIES INVOLVED: LaSalle Unit 2 DOCKET NO: 05000374

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(i) & 50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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Supervisor, Extension 2243

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: EL COMPONENT: XFMR MANUFACTURER: G082
X EF MO G083
X AD V L200
REPORTABLE NPRDS: Y
Y
Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On September 14, 1993 Unit 1 was in Operational Condition 1 (RUN) at 100% power. At 1147 hours the System Auxiliary Transformer (SAT) experienced a differential current auto-trip due to water intrusion in the 4.1 kV ductwork and a fast transfer of loads to the Unit Auxiliary Transformer (UAT) occurred. During the bus transient, the 4.1 kV bus experienced lower than normal voltage. This low voltage condition caused the Feedwater Control System for the 1B Turbine Driven Reactor Feed Pump (TDRFP) to lock up. This caused speed and flow to decrease to zero. The 1A TDRFP was unable to make up the loss in flow, and the reactor scrammed on low reactor water level. Following the Scram the UAT was lost when the Generator separated from the Grid. This resulted in a loss of

offsite power. All three diesel generators auto started and picked up the busses.

Reactor water level was restored and controlled initially by Reactor Core Isolation Cooling (RCIC), and later by Low Pressure Core Spray (LPCS). Reactor pressure was controlled by RCIC and the Safety Relief Valves (SRV). Due to a failure of the 1B Reactor Protection System (RPS) Motor Generator isolations were unable to be recovered initially. The Unit was later placed in Cold Shutdown.

This event is being reported pursuant to:
1. 10CFR50.73(a)(2)(i)(B) due to not meeting the limiting condition of operation (LCO) of Technical Specification 3.4.4.
2. 10CFR50.73(a)(2)(i)(C) due to deviating from plants Technical Specifications per 50.54(x) when LaSalle General Plant Abnormal Procedures (LGA) LGA-CM-01 and LGA-VP-01 were invoked.
3. 10CFR50.73(a)(2)(iv) due to the actuation of an Engineered Safety Feature System (ESF) and the automatic actuation of RPS.

END OF ABSTRACT

PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].

A. CONDITION PRIOR TO EVENT

Unit(s): 1 Event Date: 9/14/93 Event Time: 1147 Hours

Reactor Mode(s): 1 Mode(s) Name: Run Power Level(s): 100

B. DESCRIPTION OF EVENT

On September 14, 1993 Unit 1 was in Operational condition 1 (Run) at 100% power. At 11:47 hours a fault occurred on the Unit 1 System Auxiliary Transformer (SAT, MP) [EL]. An automatic fast transfer of loads from the SAT to the Unit Auxiliary Transformer (UAT) occurred, as designed. During the fault and bus transfer several events occurred:

(1) The 1A and 1B Turbine Driven Reactor Feed Pumps (TDRFP, FW)

[SJ] were on line. Control power was momentarily lost to the Feedwater Control System (FW) [JK]. The 1B TDRFP lost its control signal causing its speed (and flow) to decrease. The 1A TDRFP flow increased but was unable to increase enough to maintain reactor water level above the low level scram setpoint of 12.5 inches. A Unit 1 scram occurred at 19 seconds after the loss of the SAT due to low reactor water level. After the scram, reactor water level was rapidly recovered, resulting in a high water level trip (55.5 inches) of the TDRFP's and the Main Turbine resulting in a Main Generator (TG) [TB] trip on reverse power at 11:49.

(2) The Division 3 Diesel Generator (DG) [EK] started as designed on undervoltage and loaded onto its bus. Division 3 does not have a power feed from the UAT

(3) The Reactor Building Ventilation Dampers (VR) [VA] closed as well as some Primary Containment (PC)[JM] Isolation Valves as expected. A momentary loss of voltage will cause this to occur.

(4) The Unit 1 Station Air (SA) [LF] Compressor tripped as expected. This was also due to momentary drop in control power voltage.

(5) Reactor Water Cleanup (RWCU, RT) [CE] tripped due to isolation valve closure. This was caused by momentary drop in instrument power voltage.

See Attachment A for a sequence of events.

The SAT supplies power to the station from the grid and the UAT supplies power to the station from the Main Generator. With the loss of the SAT due to the fault and the Main Generator trip, Unit 1 was in a Loss of Off-site Power (LOOP) condition. The other two emergency Diesel Generators for Unit 1 auto-started on undervoltage and loaded onto their respective busses. This returned power to Unit 1 emergency busses. Additionally, the required second offsite power source to Unit 1 was available from the Unit 2 cross-tie breakers and was energized at 12:57 for Division 2 and 13:04 for Division 1 to

B. DESCRIPTION OF EVENT CONTINUED

allow unloading and securing the Diesel Generators. Additional 4.1 kV busses which could be powered from the DG or Unit 2 were energized as needed during the period when the UAT and SAT were de-energized.

LaSalle SAT and UAT provide 4.1 kV and 6.9 kV for station loads. The Emergency DG and the other Unit only supply 4.1 kV power. Therefore, the 6.9 kV busses remained de-energized. The 6.9 kV busses supply power to balance-of-plant equipment. Unit 2 was in a refueling outage at the time and receiving power from its SAT which was unaffected by this transient.

Upon the loss of the UAT additional events occurred:

(1) The Reactor Protection System (RPS, RP) [EF] and the Primary Containment Isolation System (PCIS, PC) [NH] logic initiated due to the loss of RPS bus power. The major effects were: Main Steam Isolation Valves (MSIV, MS) [SB] closure which removed the Main Condenser as the heat sink, a Primary Containment chiller isolation, a loss of low pressure Drywell Instrument Nitrogen (IN) [LE] and a Standby Gas Treatment (SBGT, VG) [BH] initiation for both trains. The RPS busses were re-energized from their respective Motor Generator (MG) sets to enable isolation recovery, but at 12:17 (approximately 30 minutes into the event), the 1B RPS MG set tripped. The alternate RPS power supply is from a 6.9 kV bus which cannot be fed from the DG or Unit 2. This resulted in the inability to easily recover from the isolation. Subsequently, at 19:07 a temporary power feed was installed for the B RPS bus which enabled isolation recovery.

(2) The Unit 2 Station Air Compressor tripped due to the loss of Unit 1 Turbine Building Closed Cooling Water (TBCCW, WT) [KB]. Unit 1 TBCCW was cross-tied supplying cooling water to the Unit 2 Station Air Compressor. Unit 1 TBCCW Pumps are powered from electrical switchgear which are supplied from 6.9 kV busses. The Unit 1 and Unit 0 Station Air Compressors became unavailable because their control power is supplied by Unit 1 6.9 kV busses.

This caused a low Instrument Air (IA) pressure condition on both Units and allowed air operated valves to go to their failed positions. The Unit 2 Scram Discharge Volume (SDV) vents and drains closed causing a Scram signal to be generated on Unit 2. Unit 2 was in the refuel mode with all control rods fully inserted and the reactor core partially unloaded prior to the event. IA was restored at approximately 14:04.

(3) The Unit 1 and 2 Fuel Pool Cooling Systems (FC) [DA] were lost due to a loss of filter/demin control power which is supplied from a Unit 1 6.9 kV bus. A temporary power feed was established and Fuel Pool Cooling was restored at 14:54. An increase in pool temperature of less than 5 degrees occurred on Unit 2. The Unit 1 Fuel Pool did not have any fuel in it at the time.

As a conservative measure and to ensure that all available help needed was assembled, the Station Manager declared an Emergency Classification Alert condition. The proper notifications to the State and the NRC were made, as required.

Safety Relief Valves (SRV, MS) [SB] were used to control reactor pressure and Reactor Core Isolation Cooling (RCIC, RI) [BN] was used for level and pressure control. The 'K' SRV opened first, however it is not in the first group of lowest pressure SRVs. Due to the loss of the low pressure Drywell Instrument nitrogen, several Automatic Depressurization System (ADS, MS) [SB] SRVs were operated using the installed backup High pressure bottled nitrogen supply. At 17:11 Low Pressure Core spray (LPCS, LP) [BM] was started for reactor water level control. At this time reactor pressure had decreased to the point where LPCS could inject.

B. DESCRIPTION OF EVENT CONTINUED

Residual Heat Removal (RHR, RH) [BO] Shutdown Cooling was established at 04:59 on September 15, 1993 and Unit 1 achieved Cold Shutdown at 11:50. At 15:15, the UAT was re-energized and at 16:56, all busses were re-energized by backfeeding through the UAT. The Emergency Plan "Alert" classification was terminated at 16:48.

Other component failures, indication problems, or items of note that occurred during the transient are:

(1) 'E' SRV failed to fully open. 'D' SRV would not open. Other SRVs had position indication problems.

(2) The RCIC and LPCS Injection Check Valves failed to indicate fully closed after injection was secured.

(3) Reactor coolant samples required due to the loss of the

Continuous Conductivity Monitor were not able to be taken within the required 4 hours per Technical Specification 4.4.4.c.1. The sample line for both methods of reactor water sampling was isolated due to the loss of the B RPS bus. These sample lines are used for both the continuous monitor and the grab sample. Technical Specification 4.4.4.c.1 requires that a grab sample be taken within 4 hours of the loss of the continuous monitor. The continuous monitor was lost at 11:49 and a sample was taken at 20:00 which exceeded the 4 hour limit. Sample results showed no fuel failure indications.

(4) Reactor coolant samples required for a 15% power change per Technical Specification 3.4.5 Action c were not able to be taken due to the loss of sample points. The Unit entered Hot Shutdown immediately following the Scram and the MSIV closed on the PCIS isolation. This met the requirements for Technical Specification 3.4.5 action a.

(5) Jumpers and lifted leads were used to bypass PC isolation signals as required per LaSalle General Plant Abnormal (LGA) Emergency Operating Procedures (EOP). The installation of these jumpers and lifted leads required, by procedure, invoking 10CFR50.54(x).

This event is being reported pursuant to:
1. 10CFR50.73(a)(2)(i)(B) due to not meeting the limiting condition of operation (LCO) of Technical Specification 3.4.4.
2. 10CFR50.73(a)(2)(i)(C) due to deviating from plants Technical Specifications per 50.54(x) when LaSalle General Plant Abnormal Procedures (LGA) LGA-CM-01 and LGA-VP-01 were invoked.
3. 10CFR50.73(a)(2)(iv) due to the actuation of an Engineered Safety Feature System (ESF) and the automatic actuation of RPS.

C. APPARENT CAUSE OF EVENT

The actual reactor Scram on Unit 1 was caused by low (12.5") reactor water level due to the loss of the 1B TDRFP. The individual failures that led to the loss of the TDRFP and the reactor scram, along with failures that occurred after the scram, are discussed individually below.

Loss of the SAT: The Unit 1 SAT auto-tripped on differential current. This was a result of water inleakage into the bus duct through degraded ductwork joint seals. This leakage accumulated in a vertical ductwork run to a surge suppressor compartment in which a sufficient quantity of water shorted

C. APPARENT CAUSE OF EVENT CONTINUED

the bus bars. An inspection revealed water marks and corrosion which indicated that the inleakage had been occurring over an extended period of time. The root cause of the bus bar short has been attributed to inadequate preventative maintenance on the ductwork seals. A contributing factor was a design which did not include low point drains in the vertical duct run.

Loss of 1B TDRFP: Immediately following the SAT fault, voltage on busses fed from the 4.1 kV side of the SAT experienced a rapid and significant decrease. For 120 Vac equipment, voltage may have been reduced to less than 72 Vac and remained degraded for a period of at least 200 msec. Based on testing performed after the event, this voltage level would initiate a decrease in the Electric Automatic Positioner (EAP) position. The EAP Controller raises or lowers turbine speed depending on the deviation between the desired speed from reactor water level control logic and actual speed. Therefore, the EAP position continued to decrease to try to zero the deviation until the lockout of the TDRFP occurred. Since the 1B TDRFP loss of signal lockout occurs at a relatively low supply voltage, the lockout is not expected to have occurred until near the end of the 200 msec time interval.

The decrease in EAP position resulted in a reduction in the 1B TDRFP speed and flow. Based on pump head/flow curves and computer data from the time immediately prior to and after the 1B TDRFP flow reduction, an EAP induced speed reduction of the 1B TDRFP of at least 1300 rpm was needed for flow to drop to zero. This amount of speed reduction is considered reasonable given the amount of EAP motion noted during testing.

Loss of the 1B RPS MG set: The 1B RPS MG set lost power following the main generator trip. It was restarted without problems and subsequently tripped due to a motor fault 30 minutes later. The motor winding was found to have a heavy layer of dirt on both ends of the winding. The winding was found to have experienced a turn to turn short at the first coil of a phase group. The most probable cause of this failure is a current spike through the motor winding caused by a transient or switching action due to interruption of power. The motor windings were also degraded from the abrasive action of the dirt on the winding insulation.

Safety Relief Valves (SRV):

1. 'K' SRV opened prior to 'S' and 'U' which have lower setpoints. 'S' and 'U' setpoints are 1076 psig with allowable tolerances of 1069 to 1099 psig. 'K' setpoint is 1096 psig with allowable tolerances of 1089 to 1119 psig. 'K' setpoint was found at 1088 psig, 1 psig out of tolerance, but within the range to operate before 'S' and 'U'.

2. 'D' SRV failed to open. The loss of power event resulted in a loss of low pressure drywell instrument nitrogen to the SRV accumulators, and 'D' SRV actuation was not required for over 2 hours after the loss of nitrogen. This was considered sufficient time for the accumulator to have bled down through the actuator block gasket leaks which were found. The ADS Accumulator was not affected. This non-safety related accumulator was not previously tested. The leaks were of sufficient quantity to prevent operation of this valve as the event occurred.

3. 'E' SRV showed dual indication when cycled from the Control Room. The valve stroke was determined to be 0.700", outside the acceptable value of 1". The spindle nut was found not tightened down properly with the load plate. This prevented the disk from going full open when stroked by the actuator.

C. APPARENT CAUSE OF EVENT CONTINUED

4. The 'C' and 'U' SRV position indication developed problems late in the event. It was determined that the valves did stroke full open, but due to the number of times these valves were cycled, the position indication became mis-calibrated.

Reactor Recirculation (RR) [AD] Pump Suction Valve (1B33-F023A): This valve failed to fully close when given a close signal from the Control Room. The valve was then reopened and closed normally when given a second close signal. Testing revealed that the 1B33-F023A problem was related to either torque switch or seal-in contacts, or actuator lubrication. It is believed that the valve may have experienced a minor internal loading problem.

RCIC and LPCS Check Valve Position Indication Problems:

1. The LPCS Check Valve (1E21-F006) did not indicate full closed

following shut down of the LPCS System. LPCS was used for vessel level control following the scram. The check valve was found in the full closed position, but the closed limit switch cam was loose, due to a stripped setscrew, resulting in improper position indication.

2. The RCIC Check Valve (1E51-F065) did not indicate full closed following the shut down of the RCIC System. The check valve was found in the partially open position as shown by the local position indication. The valve was taken full closed by rotating the external limit switch cam by hand. There was no excessive binding or internal interference preventing disc movement. The check valve is the upstream member of a pair of series check valves in the injection line. It has been previously identified and accepted that in this design, the second of the two check valves may receive insufficient differential pressure (dp) to close once the first valve has seated. Follow-up has shown that the valve easily closes by hand and would have closed if subjected to a dp. Further review of the events identified that the other check valve (1E51-F066), experienced an indication problem during the event. This problem is a loose limit switch cam caused by the design of the disk.

Safety Parameter Display System (SPDS) Failure:

During the Alert, the SPDS monitor on the unit 1 Reactor Operator Desk as well as the SPDS Monitor above Control Room Panel 1H13-P603 failed. The failure was noted early during the event, though the precise time cannot be established. The SPDS monitors were available in the Unit 2 Control Room and in the Technical Support Center at all times. Other monitors in the Control Room never lost their ability to display data. This verified that the Process Computer and other monitors in the Control Room had not failed. Also, it was verified that the UPS power to the monitors in the Control Room had not failed during the event.

The SPDS Monitor above Control Room Panel 1H13-P603 was powered from Remote Lighting Cabinet (RLC) 12. A field walkdown to determine the power source of the SPDS Monitor on the Unit 1 Reactor Operator Desk, determined that it was also being powered from an RLC circuit, though the exact RLC circuit was not verified.

All RLC circuits are powered from non-safety related busses. During the Loss of Off-site Power, the non-safety busses on Unit 1 would have load shed to allow the DGs to come online. The non-safety busses were not restored until the busses were manually restored. Control Room Personnel noted that the SPDS Monitors were operating as usual a few hours into the event.

D. SAFETY ANALYSIS OF EVENT

Reactor water level dropped to below the scram point 12.5" (level 3) but did not challenge the High Pressure Core Spray (HPCS)/RCIC initiation point of -50 inches (level 2). Reactor pressure increased to the automatic opening setpoint of an SRV in the relief mode (approximately 1070 psig), and pressure was controlled using RCIC and manually operating SRV'S. This is the normal method of level/pressure control following a scram with a MSIV closure condition. The suppression pool temperature increased to approximately 124 degrees and level to approximately +3.5 feet due to the steam discharge into the Suppression Pool from RCIC and SRV's.

A LOOP due to a SAT fault is described in Updated Final Safety Analysis Review (UFSAR) section 15.2.6 as a moderate frequency event. This event did not significantly differ from that described in the UFSAR.

A Confirmatory Action Letter (CAL) was issued after this event.

E. CORRECTIVE ACTIONS

On-site review 93-048 was initiated and approved to address issues related to this event. A summary of the corrective actions is provided below. The immediate actions taken after the scram are identified in the Description of Events.

Loss of the SAT: Oil samples of the SAT were taken for analysis to determine if damage occurred to the SAT. A meggar of the transformer and a transformer turns ratio test was performed. The 4.1 kV bus connections were disassembled, cleaned, and reassembled. The 6.9 kV bus bars were wiped down and connections retorqued. Both 4.1 kV and 6.9 kV bus duct enclosures were resealed and the bottom cover filter drain/vents replaced. Holes were drilled in the bus duct channel supports to prevent the buildup of water above the sealed connections to the internal insulator supports. The transformer low side surge suppressors of both the 4.1 kV and 6.9 kV were permanently removed. Inspection of the duct sealing tape was completed. Procedures LEP-AP-101/201 for Transformer Bus Duct

Inspections will be reviewed for clarifications and enhancements. Action Item Record (AIR) 373-240-93-04828 will track this.

Loss of 1B TDRFP: Testing revealed that the TDRFP's operated as designed as a result of the sudden voltage transient. Dahl (Manufacturer) EAP testing on the 1B TDRFP determined that a low voltage level would increase the EAP position. The new Lovejoy TDRFP Control System Modification for Unit 1, which is planned for installation during L1R06, will be evaluated for the impact of the same voltage transient. AIR 373-240-93-04801 will track this evaluation. Unit 2 TDRFP Control System is powered by an UPS and would not have been susceptible to this type of event.

1B RPS MG set: The 1B RPS MG motor was sent out for refurbishing and was replaced. The 1A motor was also sent out for cleaning and revarnishing of the windings. AIR 373-240-93-04808 was issued to revise the cleaning frequency and method of cleaning. Unit 2 RPS MG sets 2A and 2B were also inspected and found to be in an acceptable condition.

SRVs: One of 'K' SRV pressure switches was recalibrated within its tolerance. The 3 other 'K' SRV pressure switches were found within tolerance. This single drift out-of-tolerance condition is acceptable. 'D' SRV air leaks were repaired and the valve was successfully stroked. The other SRVs were tested satisfactorily. A periodic leak test of the non-ADS SRV accumulators will be performed as preventive maintenance. AIR 373-240-93-04805 will track this action.

E. CORRECTIVE ACTIONS CONTINUED

The 'U' SRV Lift Indicating Switch Assembly (LISA) was replaced, and both the 'U' and 'C' SRV LISAs were recalibrated and restroked successfully. The loose spindle nut on 'E' SRV was tightened to make contact with the loading plate and backed off one turn per vendor procedure. The other SRVs were inspected and found to be acceptable. The method of tightening the spindle nuts was revised to ensure the loading plate was secured prior to locking the spindle nut in place. AIR 373-240-93-04802 will track this item.

1B33-F023A: A current signature trace was performed and showed no new signs of valve degradation. The torque switch, which was found dirty and tarnished, was cleaned and raised to its maximum allowable setting. In addition, the valve stem and anti-rotation device were lubricated. A procedure deficiency was written to change the "preferred" valve to shut during Shutdown Cooling (SDC) operation from the 1(2)B33-F023A/B suction valve to the 1(2)B33-F067A/B discharge valve. 1B33-F023A internals will be inspected for wear or damage during L1R06. AIR 373-240-93-04829 will track this inspection.

RCIC and LPCS check valve position indication: Due to the problems with the cam set screws, all RCIC, RHR, and Emergency Core Cooling System (ECCS) Check Valves Cams were inspected. This inspection included replacing all the limit switch set screws, cleaning the internal threads (or drilling a new tap hole if the hole was stripped), and cleaning of the cam shaft with an emery cloth prior to tightening the new setscrews. AIR 373-240-93-04612 includes the inspection of both Units.

SPDS: The SPDS Monitor above Control Room Panel 1H13-P603 was rewired to be powered off an UPS Power Source. The SPDS Monitor at the Unit 1 Reactor Operators Desk was rewired and is being powered from an UPS power source. Unit 2 will be rewired prior to Unit 2 Start-Up under Modification P01-0-90-008.

Additionally, the Fuel Pool Cooling control design will be reviewed and the corrective actions will be reviewed for applicability to Unit 2. AIR 373-240-93-04815 will track this review.

F. PREVIOUS EVENTS

LER NUMBER TITLE

373/87-014-00 Reactor Scram Due to Transformer 141
Differential Current Trip

374/90-007-01 Loss of System Auxiliary Transformer Caused by a
Fire Protection Deluge of the Transformer Due to
a Short in the Deluge Manual Pull Station Switch

374/92-012-00 Reactor Scram Due to a Main Turbine Trip Caused
by a Thrust Bearing Wear Detector Signal

373/82-007-03L-0 RCIC Testable Check Valve Indication Failure

373/91-006-00 Reactor Scram On Low Reactor Vessel Level Due to
Loss of "A" Turbine Driven Reactor Feedwater
Pump Caused by Control Valve Closure

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G. COMPONENT FAILURE DATA

MANUFACTURER NOMENCLATURE MODEL NUMBER MFG PART NUMBER

General Electric System Auxiliary N/A Serial #K-547285
Transformer

General Electric Motor 5K326AN2608P N/A

Limitorque Valve SMB-0 N/A

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ATTACHMENT A

9/14/93

11:47 33 Unit 1 System Aux Transformer (SAT) lockout(trip) due to low (4160 Kv) bus duct fault. This causes voltage drop on transformer output.

Fast transfer of busses 152, 142Y, 142X, and 141Y from the SAT to the Unit Aux Transformer (UAT) occurs as expected.

0A diesel fire pump (DFP) starts due to transformer deluge.

B Reactor Recirc Flow Control Hydraulic Power Unit (HPU) isolation valves close due to momentary power loss.

U1 Service Air Compressor (SAC) trips due to momentary loss of control power.

Division 3(HPCS) SWGR 143 losses power. There is no UAT feed to Division 3.

Reactor Building Ventilation (VR) secondary containment dampers close.

37 Division 3 Diesel Generator (DG) running due to Bus 143 undervoltage

41 Reactor low level (level 4, +31.5") alarm

42 RWCU trips

44 Division 3 bus 143 energized by the Division 3 (1B) DG

52 Reactor SCRAM low water level (level 3, +12.5"). This is an LGA-01 (EOP) Entry Condition.

Reactor recirc (RR) pumps downshift to slow speed at reactor level 3 as designed.

11:48 25 Operator closes 1B Turbine Driven Reactor Feed Pump 1B discharge valve, 1FW010B as expected post scram action.

ATTACHMENT A CONTINUED

46 Reactor level increases above Level 3.

50 Operator closes 1A Turbine Driven Reactor Feed Pump 1A discharge valve, 1FW010A as an expected post scram action.

Motor Driven Reactor Feedwater Pump (MDRFP) auto starts as expected.

56 Reactor level increases above Level 4.

11:49 00 Reactor level increases above high level alarm (level 7, +41.5")

04 Reactor level increases above high level main turbine and feedwater pump trip point (Level 8, +55.5")

05 Sequence of Events Recorder memory FULL. This occurs when a large number of alarms are received in a short period of time.

06 Main Turbine trips due to high reactor level 8.

MDRFP trips due to high reactor level 8.

12 Main Generator trips due to reverse power as expected.

All busses lose power due to loss of the Unit Auxiliary Transformer (UAT) which is powered directly from the main generator.

RPS busses lose power due to loss of power to MG set.

0 Diesel Generator ENERGIZES Bus 141Y, 1A Diesel Generator ENERGIZES Bus 142Y.

MSIVs close due to loss of RPS busses.

11:50 Primary containment coolers (VP) chilled water isolation valves isolate due to loss of RPS busses.

A RHR Service Water pumps started manually for suppression pool cooling.

ATTACHMENT A CONTINUED

Drywell Instrument Nitrogen (IN) is lost due to closure of containment isolation valves due to loss of RPS busses.

11:51 Both SBGT fans start due to loss of RPS busses.

A RHR started manually for suppression pool cooling.

11:52 Reactor high pressure alarm occurs (1020 psig)

11:53 K Safety/relief valve (SRV) opened and closed on pressure. This is not the expected first SRV to open on pressure.

A/B RPS MG sets restarted.

U SRV opened manually.

11:54 S SRV opened manually.
Low-low set (LLS) initiates due to two SRVs open as expected.

Reactor high pressure alarm clears (1020 psig)

1B RPS bus re-energized

11:55 1A RPS bus re-energized

11:56 Instrument air (IA) receiver air pressure low alarm received.

11:57 Suppression Pool temperature high alarm (105 degrees) This is an LGA-03 entry condition.

11:58 S SRV closed

11:58 U SRV auto closed by LLS logic. With the SRVs closed, Reactor Level shrinks due to the collapse of voids below level 8 trip

MCC 134Y 480vac normal (From 142Y ,4.16KV bus)

RCIC manually started for injection into the core

ATTACHMENT A CONTINUED

HPCS pump manually started to place a load on 1B DG. HPCS placed in full flow test. Division 3 bus has no designed crosstie.

12:00(approx.) LGA-CM-01 jumpers installed per LGA-03 (EOP)

12:02 Suppression Pool temperature reaches 110F due to SRV's and RCIC.

12:03 B RHR Service Water manually started to support suppression pool cooling.

12:04 B RHR pump was placed in suppression pool cooling.

12:06 The Main Condenser low vacuum alarm occurs.

12:08 Primary Containment Pressure increases above 1.0 psig.

RCIC placed in full flow test and injection is secured.

12:09 Drywell air temperature reaches 135F. Another LGA-03 entry condition.

12:11 The 1B VP (primary containment chillers) loop isolation valves opened.

12:12 The 1A VP loop isolation valves opened.

12:13 Restored Bus 138 to normal manually.

12:14 Primary Containment pressure drops below 1.0 psig.

12:17 B RPS Motor Generator Set tripped the second and final time from motor fault.

12:20 Power restored to busses 131X and 131Y Manually.

SRV cycling in alphabetical sequence continues throughout the event to control/reduce reactor pressure. Reactor level shrink and swell accompany the SRV cycles. RCIC is used to control reactor level and pressure.

ATTACHMENT A CONTINUED

12:20(approx.) LGA-VP-01 jumpers installed to allow Drywell Chiller isolation to be reset per LGA-03.

12:25 NARS phone call made to report the Alert Condition. EAL 9.C. was cited.

12:47 ENS Notification Made

12:51 Bus 137X/Y Energized manually.

12:57 Service Water (WS) Pump Discharge Pressure Normal (indicative of WS Pumps back on line.

12:57 Division 1 4.1 kV bus crosstied to Unit 2, 0 Diesel Generator Shutdown, placed in Standby.

13:04 Division 2 4.1 kV bus crosstied to Unit 2, 1A Diesel Generator Shutdown, placed in Standby, Unit 2 Station Air Compressor Started, Unit 2 Turbine Building Closed Cooling Water System Started to support Air Compressor Operation.

13:18 C suppression chamber to drywell vacuum breaker opens, as expected due to SRV and RCIC adding inventory to suppression pool.

13:35 LGA-02 (EOP) Entered due to High Main Steam Tunnel Temperature.

13:45 Control transferred to TSC.

14:02 Unsuccessful attempt to open D SRV.

14:04 Instrument Air pressure Normal.

14:16 Reactor pressure at 500 psig.

RCIC Check valves appear to have not full closed.

14:54 Temporary power to fuel pool cooling (FC). Unit 2 FC system started.

15:47 Started Turbine Building Ventilation

ATTACHMENT A CONTINUED

17:04 to 19:17 Using ADS high pressure air to open SRVs.

17:08 Primary Containment Pressure High alarm, 1 psig.

17:11 LPCS Pump ON, and is now used for level control. RCIC is used to a lesser extent from here on. LPCS Injection Valve opened and closed to maintain level.

19:07 Temporary Feed Established to B RPS Bus from Alternate Feed. Isolation can be reset.

19:23 Drywell instrument air crosstie to station instrument air established. Regulator Supply alarm clears

19:26 A/B/C/D MSIV Accumulators Normal Pressure. Instrument air established to drywell.

19:50 RWCU isolation valves opened.

19:52 1G33-F034 closed

19:53 RCIC Injection Valve closed and RCIC Injection check valve 1E51-F066 Closed. This is the final use of RCIC, and the 1E51-F065 doesn't full close.

20:07 Preparing to start RWCU as indicated by valve manipulations.

20:10 RWCU started per Operations Director log.

20:11 LPCS Injection Valve Open - Indications of testable check valve indication problems.

20:25 RCIC Steam Pressure Low, reactor pressure has dropped to approximately 57 psig.

21:25 CRD Pump on, suction lined up to Condenser Hotwell

21:27 Cycled Vacuum Breakers per surveillance due to SRV usage.

21:44 SCRAM Reset

ATTACHMENT A CONTINUED

22:43 Suppression Pool Temperature Drops to less than 110 degrees F

22:44 A RHR Pump shutdown to prepare A RHR system for Shutdown Cooling.

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00:44 Shutdown Cooling Line Temperature high alarm received (indicative of line warm up in preparation to establish shutdown cooling.)

04:40 1B33-F023A fails to close on first attempt, closes on second attempt.

04:59 A RHR Pump ON - Shutdown Cooling Established

09:48 Bulk Suppression Pool Temperature Normal, 105 degrees F.

10:46 Suppression Pool Bulk Temperature Normal Div 2

11:03 Suppression Pool Bulk Temperature Normal Div 1

11:50 Operating Condition 4, Cold Shutdown is reached. LGA-01 is exited.

13:20 Unit 2 Reactor Building Ventilation is started.

13:22 Unit 1 Reactor Building Ventilation is started.

13:30 Exit LGA-02

14:24 LPCS pump shutdown

15:15 UAT Energized- Power Available

16:44 Division 1 4.1 kV fed from UAT.

16:48 6.9 kv Bus 151 Picked up, along with 480 volt sub-busses

16:48 GSEP Alert Terminated and ENS notification made in this timeframe.

ATTACHMENT A CONTINUED

16:56 6.9 kv Bus 152 Picked up, along with 480 volt sub-busses

16:58 NARs for termination issued.

17:05 TSC secured.

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19:40 Exit LGA-03

Commonwealth Edison
LaSalle County Nuclear Station
2601 N. 21st. Rd.
Marseilles, Illinois 61341
Telephone 815/357-6761

October 12, 1993

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Licensee Event Report #93-015-00, Docket #050-373 is being submitted to your office in accordance with 10CFR50.73(a)(2)(i)(B), 10CFR50.73.(a)(2)(i)(C), and 10CFR50.73(a)(2)(iv).

Joseph V. Schmeltz
Acting Station Manager
LaSalle County Station

JVS/RR/grv

Enclosure

xc: Nuclear Licensing Administrator
NRC Resident Inspector
NRC Region III Administrator
INPO - Records Center
IDNS Resident Inspector

ATTACHMENT 1 TO 9310210020 PAGE 2 OF 2

Figure "Event Summary and Cause Codes" omitted.

*** END OF DOCUMENT ***